UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 28, 2018

Mr. James J. Hutto
Regulatory Affairs Director
Southern Nuclear Operating Company, Inc.
P.O. Box 1295 / Bin – 038
Birmingham, AL 35201-1295

SUBJECT: JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2 – ISSUANCE OF AMENDMENTS TO REVISE TECHNICAL SPECIFICATION 5.5.17, "CONTAINMENT LEAKAGE RATE TESTING PROGRAM" (CAC NOS. MF8844, MF8845; EPID NO. L-2016-LLA-0015)

Dear Mr. Hutto:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 217 to Renewed Facility Operating License No. NPF-2 and Amendment No. 214 to Renewed Facility Operating License No. NPF-8 for the Joseph M. Farley Nuclear Plant, Units 1 and 2, respectively. The amendments consist of changes to the technical specifications (TSs) in response to your application dated November 15, 2016, as supplemented by letters dated June 22, 2017, September 11, 2017, October 12, 2017, and February 9, 2018.

The amendments revise TS 5.5.17, "Containment Leakage Rate Testing Program." Specifically, the amendments increase the existing Type A integrated leakage rate test program test interval from 10 years to 15 years; adopt an extension of the containment isolation valve leakage testing (Type C) frequency from 60 months to 75 months; adopt the use of American National Standards Institute/American Nuclear Society (ANSI/ANS) 56.8-2002, "Containment System Leakage Testing Requirements"; and adopt a grace interval of 9 months for Type A, Type B, and Type C leakage tests, in accordance with Nuclear Energy Institute (NEI) 94-01, Revision 2-A and Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J."

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Shawn Williams

Shawn A. Williams, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-348 and 50-364

Enclosures:
1. Amendment No. 217 to NPF-2
2. Amendment No. 214 to NPF-8
3. Safety Evaluation

cc w/enclosures: Distribution via Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

ALABAMA POWER COMPANY

DOCKET NO. 50-348

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 217
Renewed License No. NPF-2

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Joseph M. Farley Nuclear Plant, Unit 1, (the facility), Renewed Facility Operating License No. NPF-2, filed by Southern Nuclear Operating Company, Inc. (the licensee), dated November 15, 2016, as supplemented by letters dated June 22, 2017, September 11, 2017, October 12, 2017, and February 9, 2018, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-2, is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 217, are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility
Operating License and
Technical Specifications

Date of Issuance: February 28, 2018

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SOUTHERN NUCLEAR OPERATING COMPANY

ALABAMA POWER COMPANY

DOCKET NO. 50-364

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 214
Renewed License No. NPF-8

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Joseph M. Farley Nuclear Plant, Unit 2, (the facility), Renewed Facility Operating License No. NPF-8, filed by Southern Nuclear Operating Company, Inc. (the licensee), dated November 15, 2016, as supplemented by letters dated June 22, 2017, September 11, 2017, October 12, 2017, and February 9, 2018, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-8 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 214, are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

Michael T. Markley, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility
Operating License
and Technical Specifications

Date of Issuance: February 28, 2018

ATTACHMENT TO JOSEPH M. FARLEY NUCLEAR PLANTS

UNITS 1 AND 2

LICENSE AMENDMENT NO. 217

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-2

DOCKET NO. 50-348

AND LICENSE AMENDMENT NO. 214

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-8

DOCKET NO. 50-364

Replace the following pages of the Renewed Facility Operating Licenses and Appendix "A" Technical Specifications (TSs) with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
License	License
NPF-2, page 4	NPF-2, page 4
NPF-8, page 3	NPF-8, page 3
TSs	TSs
5.5-14	5.5-14

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 217, are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications.

(3) Additional Conditions

The matters specified in the following conditions shall be completed to the satisfaction of the Commission within the stated time periods following the issuance of the renewed license or within the operational restrictions indicated. The removal of these conditions shall be made by an amendment to the renewed license supported by a favorable evaluation by the Commission.

a. Southern Nuclear shall not operate the reactor in Operational Modes 1 and 2 with less than three reactor coolant pumps in operation.

b. Deleted per Amendment 13

c. Deleted per Amendment 2

d. Deleted per Amendment 2

e. Deleted per Amendment 152

   Deleted per Amendment 2

f. Deleted per Amendment 158

g. Southern Nuclear shall maintain a secondary water chemistry monitoring program to inhibit steam generator tube degradation. This program shall include:

   1) Identification of a sampling schedule for the critical parameters and control points for these parameters;

   2) Identification of the procedures used to quantify parameters that are critical to control points;

   3) Identification of process sampling points;

   4) A procedure for the recording and management of data;

   5) Procedures defining corrective actions for off control point chemistry conditions; and

(2) Alabama Power Company, pursuant to Section 103 of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," to possess but not operate the facility at the designated location in Houston County, Alabama in accordance with the procedures and limitations set forth in this renewed license.

(3) Southern Nuclear, pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(4) Southern Nuclear, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) Southern Nuclear, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) Southern Nuclear, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) <u>Maximum Power Level</u>

Southern Nuclear is authorized to operate the facility at reactor core power levels not in excess of 2775 megawatts thermal.

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 214 are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications.

(3) Deleted per Amendment 144

(4) Deleted per Amendment 149

(5) Deleted per Amendment 144

5.5 Programs and Manuals

5.5.17 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of containment as required by 10 CFR 50.54 (o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 3-A, dated July 2012, and the conditions and limitations specified in NEI 94-01, Revision 2-A, dated October 2008 as modified by the following exceptions:

1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 43.8 psig.

The maximum allowable containment leakage rate, L_a, at P_a, is 0.15% of containment air weight per day.

Leakage rate acceptance criteria are:

a. Containment overall leakage rate acceptance criterion is $\leq 1.0\ L_a$. During plant startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the combined Type B and C tests, and $\leq 0.75\ L_a$ for Type A tests;

b. Air lock testing acceptance criteria are:

   1. Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

   2. For each door, leakage rate is $\leq 0.01\ L_a$ when pressurized to ≥ 10 psig.

c. During plant startup following testing in accordance with this program, the leakage rate acceptance criterion for each containment purge penetration flowpath is $\leq 0.05\ L_a$

(continued)

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 217 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-2

AND

AMENDMENT NO. 214 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-8

SOUTHERN NUCLEAR OPERATING COMPANY

JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-348 AND 50-364

1.0 INTRODUCTION

By application dated November 15, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16320A540), as supplemented by letters dated June 22, 2017 (ADAMS Accession No. ML17173A652), September 11, 2017 (ADAMS Accession No. ML17255A159), October 12, 2017 (ADAMS Accession No. ML17285B308), and February 9, 2018 (ADAMS Accession No. ML18043A175), the Southern Nuclear Operating Company, Inc., (SNC, the licensee) submitted a request to change the Joseph M. Farley Nuclear Plant (FNP or Farley), Units 1 and 2, Technical Specifications (TSs).

The amendments revise TS 5.5.17, "Containment Leakage Rate Testing Program." Specifically, the amendments increase the existing Type A integrated leakage rate test program test (ILRT) interval from 10 years to 15 years; adopt an extension of the containment isolation valve leakage testing (Type C) frequency from 60 months to 75 months; adopt the use of American National Standards Institute/American Nuclear Society (ANSI/ANS) 56.8-2002, "Containment System Leakage Testing Requirements"; and adopt a grace interval of 9 months for Type A, Type B, and Type C leakage tests, in accordance with Nuclear Energy Institute (NEI) 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated November 19, 2008 (ADAMS Accession No. ML100620847), and Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 31, 2012 (ADAMS Accession No. ML12221A202). With this change, FNP will implement the guidance of NEI 94-01, Revision 3-A and Revision 2-A, with their associated limitations and conditions, rather than those of NEI 94-01, Revision 0, July 26, 1995 "Industry Guideline for Implementing Performance Based Option of 10 CFR Part 50, Appendix J" (ADAMS Accession No. ML11327A025) that was endorsed by Regulatory Guide 1.163, Revision 0.

The supplemental letters dated June 22, 2017, September 11, 2017, October 12, 2017, and February 9, 2018, provide additional information that clarified the application, did not expand the scope of the application as originally `noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on January 3, 2017 (82 FR 161).

2.0 REGULATORY EVALUATION

2.1 System Description

The FNP Units 1 and 2 are Westinghouse design 3-loop pressurized water reactors within large, dry ambient design containments. The primary containment consists of a pre-stressed tendon and reinforced concrete vertical right cylinder topped by a shallow dome and a reinforced concrete foundation slab. A ¼ inch thick welded steel plate inside liner is attached to the inside face of the concrete. The penetrations through the primary containment and this liner, which is somewhat thicker around the penetrations, provides for leak tightness.

The primary containment provides the "leak tight" barrier against the potential uncontrolled release of fission products during a design basis loss of coolant accident (DBA-LOCA). The TS 5.5.17 identifies the primary containment allowable leakage rate (L_a) as 0.15 percent of the containment air weight per day at the calculated maximum DBA-LOCA pressure (P_a) of 43.8 psig.

2.2 Licensee Proposed Changes

The current TS 5.5.17, "Containment Leakage Rate Testing Program," states, in part:

> A program shall be established to implement the leakage rate testing of containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak Test Program," dated September 1995 [ADAMS Accession No. ML003740058], as modified by the following exception to NEI 94-01, Revision 0, "Industry Guidelines for Implementing Performance-Based Option of 10 CFR 50, Appendix J":

The proposed change will revise TS 5.5.17 to state, in part:

> A program shall be established to implement the leakage rate testing of containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in NEI 94-01, "Industry Guideline for implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 3-A, dated July 2012, and the conditions and limitations specified in NEI 94-01, Revision 2-A, dated October 2008 as modified by the following exceptions.

FNP will implement the guidance of NEI 94-01, Revision 3-A and Revision 2-A, with the associated limitations and conditions, rather than those of NEI 94-01, Revision 0, that was endorsed by Regulatory Guide 1.163, Revision 0. This is permitted by the provision in 10 CFR 50, Appendix J, Option B, Section V.B.3 regarding the TS referencing the NRC staff approved guidance document for program implementation. Guidance in NEI 94-01, Revision 3-A, provides that extension of the Type A test (ILRT) interval to 15 years be based on

two consecutive successful Type A tests (performance history) and other requirements stated in Section 9.2.3 of NEI 94-01, Revision 3-A. The basis for acceptability of extending the Type A test interval also includes implementation of robust Type B and Type C testing of the penetration barriers where most containment leakage has historically been shown to occur and are expected to continue to be the pathways for a majority of potential primary containment leakage; and of a robust containment visual inspection program where deterioration of the primary containment boundary away from penetrations can be detected and remediated before any actual significant leakage potential were to develop. Guidance in NEI 94-01, Revision 3-A, also provides that Type C test intervals may be extended to 75 months based on two consecutive successful tests (performance history) and meeting other specified conditions.

In summary, the proposed change to TS 5.5.17 would result in the following:

- An increase in the existing Type A ILRT interval from 10 years to 15 years in accordance with NEI 94-01, Revisions 3-A, and the conditions and limitations specified in NEI 94-01, Revision 2-A.
- An extension of the containment isolation valve (CIV) leakage testing (Type C) frequency from 60 months to 75 months.
- Adopting the use of American National Standards Institute/American Nuclear Society (ANSI/ANS) 56.8-2002, "Containment System Leakage Testing Requirements."
- Adopting a grace interval of 9 months for Type A, Type B, and Type C leakage tests, in accordance with NEI 94-01, Revision 3-A.

2.3 Regulatory Requirements and Guidance

FNP General Design Criterion 16, "Containment Design" states, in part:

> Reactor containment and associated systems are provided to establish an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

The regulations in Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50.54(o) requires primary reactor containments for water cooled power reactors be subject to the requirements set forth in 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." Appendix J to 10 CFR 50, includes two options: "Option A – Prescriptive Requirements," and "Option B – Performance- Based Requirements," either of which may be chosen by a licensee for meeting the requirements of Appendix J. FNP, Units 1 and 2, adopted 10 CFR 50, Appendix J, Option B for Type A ILRT, and Type B and Type C local leak rate tests (LLRT).

The testing requirements in 10 CFR Part 50, Appendix J ensure that (a) leakage through containments or systems and components penetrating containments does not exceed allowable leakage rates specified in the TS; and (b) integrity of the containment structure is maintained during the service life of the containment. Option B of Appendix J specifies performance-based requirements and criteria for preoperational and subsequent leakage rate testing of the primary containment. The requirements set forth in Appendix J are satisfied by performing a Type A test to measure the overall integrated leakage rate of the primary containment; Type B test consisting of a pneumatic test to detect and measure local leakage rates across pressure-retaining and leakage-limiting boundaries; and Type C test consisting of a pneumatic

tests to measure containment isolation valve leakage rates. Following satisfactory tests prior to initial criticality, periodic tests thereafter are required to be conducted at intervals based on the historical performance of the overall containment system (for Type A tests), and based on the safety significance and historical performance of each penetration boundary and isolation valves (for Type B and C tests) to ensure integrity of the overall containment system as a barrier to fission product release.

The leakage rate test results must not exceed the allowable leakage rate (L_a) as specified in the TS. Option B also requires that a general visual inspection of the accessible interior and exterior surfaces of the containment system, for structural deterioration which may affect the containment leak-tight integrity, must be conducted prior to each Type A test and at a periodic interval between tests.

Section V.B.3 of 10 CFR 50, Appendix J, Option B, requires that the regulatory guide or other implementation document used by a licensee to develop a performance-based leakage testing program must be included, by general reference, in the plant TSs. Furthermore, the submittal for TS revisions must contain justification, including supporting analyses, if the licensee chooses to deviate from methods approved by the NRC and endorsed in a regulatory guide.

The NRC staff's final Safety Evaluation (SE) for NEI 94-01, Revision 2, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," and Electric Power Research Institute (EPRI) Report No. 1009325, Revision 2, August 2007," 'Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," June 25, 2008 (ADAMS Accession No. ML081140105) was incorporated into NEI 94-01, Revision 2-A, November 2008. NEI 94-01, Revision 2-A, describes an NRC-approved approach for implementing the optional performance-based requirements of Option B described in 10 CFR Part 50, Appendix J, which includes provisions for extending Type A ILRT intervals to up to 15 years, and incorporates the regulatory positions stated in Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Rate Testing Program," dated September 1995 (ADAMS Accession No. ML003740058). NEI 94-01, Revision 2-A, delineates a performance-based approach for determining Type A, Type B, and Type C containment leakage rate surveillance testing frequencies. This method uses industry performance and plant-specific data and risk insights in determining the appropriate testing frequency, and also discusses the performance factors that licensees must consider in determining test intervals. NEI 94-01, Revision 2-A, includes six specific limitations and conditions listed in Section 4.1 of the SE.

The NRC's staff's final SE of NEI 94-01, Revision 3, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," June 8, 2012 (ADAMS Accession No. ML121030286) was incorporated into NEI 94-01, Revision 3-A, dated July 2012. NEI 94-01, Revision 3-A, documents the NRC's evaluation and acceptance of NEI 94-01, Revision 3, and includes two specific limitations and conditions listed in Section 4.0 of the SE.

Title 10 CFR 50.55a "Codes and Standards," contains the containment inservice inspection (ISI) program requirements that, in conjunction with the requirements of Appendix J, ensure the continued leak-tight and structural integrity of the containment during its service life.

Title 10 CFR 50.65 (a), "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," states, in part, that the licensee "...shall monitor the performance or condition of structures, systems, or components, against licensee-established goals, in a manner sufficient to provide reasonable assurance that these structures, systems, and components, as defined in paragraph (b) of this section, are capable of fulfilling their intended

functions. These goals shall be established commensurate with safety and, where practical, take into account industry-wide operating experience."

Title 10 CFR 50.36, "Technical specifications" states that the TSs include items in five specific categories. These categories include: (1) safety limits, limiting safety system settings, and limiting control settings; (2) Limiting Condition for Operations; (3) surveillance requirements; (4) design features; and (5) administrative controls. NUREG-1431, "Standard Technical Specifications - Westinghouse Plants," Revision 4.0, (ADAMS Accession No. ML12100A222) incorporated the Standard Technical Specification Task Force Traveler (TSTF)-52, Revision 3 (ADAMS Accession No. ML040400371), that includes guidance for specific changes to TS for implementation of 10 CFR 50, Appendix J, Option B.

The EPRI Report No. 1009325, Revision 2-A, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals" October 2008 (ADAMS Accession No. ML14024A045), provides a risk impact assessment for optimized ILRT intervals of up to 15 years, utilizing current industry performance data and risk-informed guidance, primarily Revision 2 of RG 1.174, "An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" May 2011 (ADAMS Accession No. ML100910006). EPRI Report No. 1009325, Revision 2-A, determined that there is very little risk associated with extension of ILRT intervals to 15 years.

The implementation document that is currently referenced in the FNP TS is RG 1.163, "Performance-Based Containment Leak-Test Program," September 1995, which endorsed TR NEI 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," July 26, 1995, as a document that provides methods acceptable to the NRC staff for complying with the provisions of Option B of 10 CFR Part 50, Appendix J, subject to four regulatory positions delineated in Section C of the RG. NEI 94-01, Revision 0, includes provisions that allow the performance-based Type A test interval to be extended to up to 10 years, based upon two consecutive successful tests.

By letter dated March 21, 2003 (ADAMS Accession No. ML030800326), the NRC issued License Amendment Nos. 159 for Unit 1 and 150 for Unit 2, respectively, to allow a revision to TS 5.5.17 for a one-time deferral of FNP Type A containments ILRT from 10 years to 15 years pursuant to 10 CFR Part 50, Appendix J, Option B. Previously, on September 3, 1996, the NRC issued License Amendment Nos. 122 for Unit 1 and 114 for Unit 2 (ADAMS Accession Legacy Package No. 9609100108), which modified the TS to reflect the licensee's implementation of 10 CFR 50, Appendix J, Option B.

3.0 <u>TECHNICAL EVALUATION</u>

3.1 Balance of Plant Staff Evaluation

In accordance with the guidance in NEI 94-01, Revision 2-A, subject to the NRC conditions and limitations, the licensee proposed to extend the containment Type A test interval from the current approved 10 years to 15 years, based on acceptable performance. This would allow the next Type A test to be performed within 15 years of the last acceptable test. The previous Type A tests were performed in April 2009 (Unit 1) and April 2010 (Unit 2). The approval of this amendment would allow the next Type A tests to be performed no later than April 2024 and April 2025 for Unit 1 and Unit 2, respectively.

In accordance with the guidance in NEI 94-01, Revision 3-A, subject to two conditions and limitations, the licensee proposes to extend the containment Type C test interval from the current approved 60 months to 75 months, with a permissible extension period of 9 months (total of 84 months) for non-routine emergent conditions, based on acceptable performance. This would allow the next Type C test to be performed within 75 months from the last test.

3.1.1 Historical Type A Test (ILRT) Results

In the November 15, 2016, licensing application request (LAR), Table 3.2.4-1, FNP-1 Type A ILRT History, the licensee presented the historical results of the FNP Unit 1 ILRT as summarized below.

Test Date	Leakage Rate (Primary Containment Percent Weight % per Day)
November, 1986	0.042
May, 1991	0.055
March, 1994	0.048
March, 2009	0.049

The ILRT performance criterion (L_a) is 0.15 weight % per day. The NEI 94-01, Revision 3-A, requirement for allowing the extended test interval is that the past two tests meet the performance criterion by showing a leakage of L_a or less.

The FNP Unit 1, 1994 and 2009 ILRT showed leakage to be about 1/3rd of the performance criterion. Thus, the NEI 94-01, Revision 3-A, requirement that the past two tests meet the performance criterion by showing a leakage of L_a or less is met for FNP Unit 1. Past testing results show ample margin and no adverse trend and thus suggest that an ILRT interval of 15 years would not result in exceeding the performance criterion for FNP Unit 1.

In LAR Table 3.2.4-2, FNP-2 Type A ILRT History, the licensee presented the historical results of the FNP Unit 2 ILRT as summarized below.

Test Date	Leakage Rate (Primary Containment Percent Weight % per Day)
November, 1987	0.064
December, 1990	0.048
March, 1995	0.111
April, 2010	0.024

The FNP Unit 2, 1995 and 2010 ILRT showed leakage to be well within the performance criterion. Thus, the NEI 94-01, Revision 3-A, requirement that the past two tests meet the performance criterion by showing a leakage of L_a or less is met for FNP Unit 2. Past testing results show ample margin and no adverse trend and thus suggest that an ILRT interval of 15 years would not result in exceeding the performance criterion for FNP Unit 2.

3.1.2 Historical Type B and Type C Test (LLRT) Results

In LAR Table 3.4.5-1 the licensee presented the historical results of the Type B and C test combined as-found minimum pathway leakage totals for FNP Unit 1 as summarized below:

Refuel Outage / Year	As-Found Minimum Pathway Leakage Rate (standard cubic centimeters per minute) (sccm)	Fraction of TS 5.5.17 Combined Type B and C Performance Criterion (0.6 L_a) in percent (%)
1R20 / 2006	20,675.84	14.68%
1R21 / 2007	7,732.66	5.49%
1R22 / 2009	16,325.76	11.59%
1R23 / 2010	18,834.77	13.37%
1R24 / 2012	12,734.72	9.00%
1R25 / 2013	17,738.61	12.59%
1R26 / 2015	17,755.86	12.61%

The performance criterion for combined Type B and Type C test total with TS 5.5.17 specified margin is 0.6 L_a as summed from the as-found minimum pathway leakage values. The licensee states in LAR Section 3.4.5 that 0.6 L_a is 140,862 standard cubic centimeters per minute (sccm).

The FNP Unit 1 combined Type B and Type C test totals since the last ILRT performance in 2009 all show substantial additional margin suggesting that that performance criteria are unlikely to be exceeded by allowing FNP Unit 1 ILRT maximum interval to be extended to 15 years or the Type C test intervals to be extended to 75 months.

In LAR Table 3.4.5-2 the licensee presented the historical results of the Type B and Type C test combined as-found minimum pathway leakage totals for FNP Unit 2 as summarized below:

Refuel Outage / Year	As-Found Minimum Pathway Leakage Rate (standard cubic centimeters per minute) (sccm)	Fraction of TS 5.5.17 Combined Type B and C Performance Criterion (0.6 L_a) in percent (%)
2R18 / 2007	31,429.70	22.31%
2R19 / 2008	11,551.67	8.20%
2R20 / 2010	15,949.27	11.32%
2R21 / 2011	16,079.54	11.42%
2R22 / 2013	21,487.82	15.25%
2R23 / 2014	9,245.2	6.56%
2R24 / 2016	33,346.84	18.94%

The FNP Unit 2 combined Type B and Type C test totals since the last ILRT performance in 2010 all show substantial additional margin suggesting that that performance criteria are unlikely to be exceeded by allowing FNP Unit 2 ILRT maximum interval to be extended to 15 years or the Type C test intervals to be extended to 75 months.

3.1.3 Conditions and Limitations in NEI 94-01, Revision 2-A

In the SE for NEI 94-01, Revision 2-A, the NRC staff concluded that the guidance in Topical Report (TR) NEI 94-01, Revision 2, is acceptable for reference by licensees proposing to amend their TS in regards to containment leakage rate testing, subject to six conditions and limitations. The requirements of NEI 94-01 stayed essentially the same from the original version through Revision 2 except that the regulatory positions of RG 1.163 were incorporated and the maximum ILRT interval extended to 15 years. The LAR Table 3.8.1-1 described the licensee response to the six conditions and limitations identified in the SE dated June 25, 2008, and the NRC staff evaluated these responses to determine whether the licensee adequately addressed these conditions.

NEI 94-01, Revision 2-A, Condition 1

> For calculating the Type A leakage rate, the licensee should use the definition in the NEI TR 94-01, Revision 2, in lieu of that in ANSI/ANS- 56.8-2002. (Refer to SE Section 3.1.1.1).

NRC Staff Evaluation

The licensee stated in the LAR that they would be using the definition in NEI 94-01, Revision 2-A, Section 5.0, which remained the same in NEI 94-01, Revision 3-A. This definition is acceptable, therefore, the NRC staff concludes that the licensee has addressed Condition 1 sufficiently.

NEI 94-01, Revision 2-A, Condition 2

> The licensee submits a schedule of containment inspections to be performed prior to and between Type A tests. (Refer to SE Section 3.1.1.3).

NRC Staff Evaluation

The LAR provided the current schedule of containment inspections in the LAR, Tables 3.4.2-1 through 3.4.2-15 for the FNP Unit 1 and 2 implementation of American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel Code Section XI, Subsections IWE, "Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Plants", and IWL, "Requirements for Class CC Concrete Components of Light-Water Cooled Plants". Therefore, the NRC staff concludes that the licensee has addressed Condition 2 sufficiently.

NEI 94-01, Revision 2-A, Condition 3

> The licensee addresses the areas of the containment structure potentially subjected to degradation. (Refer to SE Section 3.1.3).

NRC Staff Evaluation

In the November 15, 2016, LAR, the licensee described in Section 3.4.2 "Containment Inservice Inspection Program," Section 3.5 "Operating Experience," and Section 3.6, "Primary Containment Operating Experience since Completion of Last ILRT in 2009 and 2010" FNP experience regarding degradation found in past inspections and how inaccessible areas would be assessed for degradation potential.

In Section 3.4.2, the licensee states:

> Augmented Examination and Supplemental Examinations
>
> Areas meeting the selection criteria of IWE-1240 will receive augmented examination. Areas may be added as conditions warrant and also may be removed from Augmented Examination status as permitted by IWE-2420(c). Results of augmented examinations in accessible areas will be evaluated for their potential to affect inaccessible areas. The results of this evaluation will determine the scope and extent of augmented examinations in the inaccessible areas.
>
> IWE-1241 requires areas likely to experience accelerated degradation and aging to be subject to the augmented examinations specified in Table IWE-2500-1, Category E-C. Item E4.10 requires detailed visual examination (VT-1) and volumetric examination (UT) of areas exhibiting signs of accelerated degradation of areas having conditions, which could cause accelerated degradation. Currently, there are no areas exhibiting these conditions.

The licensee discussed past experience with conditions that have potential for accelerated degradation, specifically with deteriorated moisture barriers and water at concrete to liner interfaces. Some conditions were not in the containment ISI program that should have been (i.e., test connection accesses to the liner weld channels under the concrete floor slab) that were added subsequently to the containment ISI program and augmented inspected. The licensee stated that the potential for similar conditions to inaccessible containment boundary surface areas were assessed. The licensee also stated that currently there are no areas exhibiting signs of accelerated degradation or areas having conditions that could cause accelerated degradation.

The NRC staff concludes that the licensee has addressed Condition 3 sufficiently.

<u>NEI 94-01, Revision 2-A, Condition 4</u>

> The licensee addresses any tests and inspections performed following major modifications to the containment structure, as applicable. (Refer to SE Section 3.1.4)."

NRC Staff Evaluation

The LAR indicated that steam generator replacements at FNP involved use of the containment equipment hatches and not with temporary openings. The LAR also indicated that there are no major modifications planned (affecting the containments). Therefore, the NRC staff concludes that the licensee has addressed Condition 4 sufficently.

<u>NEI 94-01, Revision 2-A, Condition 5</u>

> The normal Type A test interval should be less than 15 years. If a licensee has to utilize the provision of Section 9.1 of NEI TR 94-01, Revision 2, related to extending the ILRT interval beyond 15 years, the licensee must demonstrate to the NRC staff that it is an unforeseen emergent condition. (Refer to SE Section 3.1.1.2)."

NRC Staff Evaluation

The licensee's response acknowledges and accepts this Condition. The licensee is not proposing to extend the ILRT interval beyond 15 years; and thus, the licensee's response is acceptable to the NRC staff.

<u>NEI 94-01, Revision 2-A, Condition 6</u>

> For plants licensed under 10 CFR Part 52, applications requesting a permanent extension of the ILRT surveillance interval to 15 years should be deferred until after the construction and testing of containments for that design have been completed and applicants have confirmed the applicability of NEI TR 94-01, Revision 2, and EPRI Report No. 1009325, Revision 2, including the use of past ILRT data.

NRC Staff Evaluation

This condition is not applicable to FNP, Units 1 and 2, because they were not licensed under 10 CFR Part 52.

3.1.4 Conditions and Limitations in NEI 94-01, Revision 3-A

In the SE for NEI 94-01, Revision 3-A, the NRC staff concluded that the guidance in TR NEI 94-01, Revision 3-A, is acceptable for reference by licensees proposing to amend their TS in regards to containment leakage rate testing, subject to two conditions and limitations.

<u>NEI 94-01, Revision 3-A, Condition 1</u>

> NEI TR 94-01, Revision 3, is requesting that the allowable extended interval for Type C LLRTs be increased to 75 months, with a permissible extension (for non-routine emergent conditions) of nine months (84 months total). The staff is allowing the extended interval for Type C LLRTs be increased to 75 months with the requirement that a licensee's post-outage report include the margin between the Type B and Type C leakage rate summation and its regulatory limit. In addition, a corrective action plan shall be developed to restore the margin to an acceptable level. The staff is also allowing the non-routine emergent extension out to 84-months as applied to Type C valves at a site, with some exceptions that must be detailed in NEI 94-01, Revision 3. At no time shall an extension be allowed for Type C valves that are restricted categorically (e.g. BWR MSIVs), and those valves with a history of leakage, any valves held to either a less than maximum interval or to the base refueling cycle interval. Only non-routine emergent conditions allow an extension to 84 months.

<u>NEI 94-01, Revision 3-A, Condition 2</u>

> When routinely scheduling any LLRT valve interval beyond 60-months and up to 75-months, the primary containment leakage rate testing program trending or monitoring must include an estimate of the amount of understatement in the Type B & C total, and must be included in the post-outage report. The report must include the reasoning and determination of the acceptability of the

extensions, demonstrating that the LLRT totals calculated represent the actual leakage potential of the penetrations.

NRC Staff Evaluation

The licensee indicated in the LAR that the FNP post-outage reports will include the margin between the Type B and Type C minimum pathway leak rate summation value adjusted for understatement and the acceptance criterion. Should the Type B and Type C combined totals exceed an administrative limit of 0.5 L_a but be less than the TS acceptance value (performance criterion) of 0.6 L_a, then an analysis will be performed and a corrective action plan prepared to restore and maintain the leakage summation margin to less than the administrative limit. The LAR also stated that FNP will apply the 9-month grace period only to eligible Type C tested components and only for non-routine emergent conditions. The licensee acknowledges these two conditions and the likelihood that longer test intervals would increase the understatement of actual leakage potential given the method by which the totals are calculated, and will assign additional margin for monitoring acceptability of results via administrative limits and understatement contribution adjustments. Therefore, the NRC staff concludes that the licensee has addressed Conditions 1 and 2 of NEI 94-01 Revision 3-A sufficiently.

3.1.5 NRC Staff Conclusion

Based on the above, the NRC staff concludes that the requested change is within the regulatory requirements and guidance as described in Section 2.0 of this SE. The NRC staff concludes that the licensee has adequately implemented its primary containment leakage rate testing program consisting of ILRT and LLRT. The results of the recent ILRTs and the LLRT (Type B and Type C tests) combined totals demonstrate acceptable performance and support a conclusion that the structural and leak-tight integrity of the primary containment vessel is adequately managed and will continue to be periodically monitored and managed effectively. The NRC staff concludes that the licensee has addressed the NRC conditions and limitations to demonstrate acceptability of adopting TR NEI 94-01, Revision 3-A, and the limitations and conditions identified in the staff SE incorporated in TR NEI 94-01, Revision 2-A. Therefore, the NRC staff concludes that the proposed changes to FNP TS 5.5.17 regarding the primary containment leakage rate testing program are acceptable.

3.2 Structural Engineering Staff Evaluation

3.2.1 Extension of Type A Test Interval from 10 Years to 15 Years

As described in the FNP Updated Final Safety Analysis Report (ADAMS Package Accession No. ML17117A380), Section 3.8.1, "Concrete Containment," the primary containment for each of the FNP units is a pre-stressed reinforced concrete cylindrical structure with a shallow dome roof and reinforced concrete foundation slab. A ¼ inch-thick welded steel liner is attached to the inside face of the concrete. The floor liner is installed on top of the foundation slab and is covered with concrete. The cylindrical portion of the containment is pre-stressed by a post-tensioning system composed of horizontal and vertical tendons. Inside the containment, the reactor and other nuclear steam supply system components are shielded with concrete. A vent stack is attached to the outside of the containment and extends above the top of the containment dome. Access to portions of the containment during power operation is permissible. The containment, in conjunction with engineered safety features, is designed to withstand the internal pressure and coincident temperature resulting from the energy release of a loss-of-coolant accident. The leak-tight integrity of the penetrations and isolation valves are

verified through local leak rate testing, Type B and Type C tests, and the overall leak-tight integrity and structural integrity of the primary containment is verified through an ILRT Type A test, as required by 10 CFR 50, Appendix J. The leakage rate testing requirements of 10 CFR 50, Appendix J, Option B (Type A, Type B and Type C Tests) and the containment ISI requirements mandated by
10 CFR 50.55a, together, ensure the continued leak-tight and structural integrity of the containment during its service life.

3.2.2 Historical Plant-Specific Containment Leakage Testing Program Results

As provided in Section 3.2.4 of the LAR and FNP TS 5.5.17, the maximum allowable containment leakage rate, L_a, is 0.15 percent of containment air weight per day at the peak calculated containment internal pressure for design basis loss-of-coolant accident, P_a. The containment overall leakage rate acceptance criterion is less than or equal to L_a. During plant startup following testing in accordance with the licensee's program, the leakage rate acceptance criteria is less than or equal to 0.75 L_a for Type A tests. The current FNP TSs require Type A testing in accordance with RG 1.163 which endorses the methodology for complying with Option B. The licensee stated that the performance leakage rates are calculated in accordance with NEI 94-01, Section 9.1.1, for Type A testing. Option B also requires that a general visual inspection be performed on the accessible interior and exterior surfaces of the containment structure for structural deterioration which may affect the containment leak-tight integrity prior to each Type A test, and at a periodic interval between tests based on the performance of the containment system.

In Section 3.2.4 of the LAR, the licensee provided Tables 3.2.4-1 (Unit 1) and 3.2.4-2 (Unit 2) which provided the past periodic Type A ILRT leakage rate test results. The leak rate results of the last two Unit 1 Type A tests performed on March 1994 (RFO 1R21) and April 2009 (RFO 1R22) were 0.048 percent and 0.049 percent of containment air weight per day, respectively. The leak rate results of the last two Unit 2 Type A tests performed on March 1995 (RFO 2R19) and April 2010 (RFO 2R20) were 0.111 percent and 0.024 percent of containment air weight per day, respectively. The licensee stated that the resulting calculated value for Unit 1 is 3 times less than the Maximum Allowable Leakage Rate, L_a, and is 2.4 times less than the acceptance criteria for the ILRT. For Unit 2, the resulting calculated value is 6.5 times less than the Maximum Allowable Leakage Rate, L_a, and is 4.8 times less than the acceptance criteria for the ILRT.

NRC Staff Evaluation

The NRC staff finds that the resulting calculated leakage indicates that a significant margin is left between the actual calculated leakage, and the allowable values which indicates that containment integrity is maintained for both units.

3.2.3 Containment ISI Program (ASME Section XI, Subsections IWE and IWL)

3.2.3.1 ASME Section XI, Subsection IWE

In Sections 3.4.2 and 3.4.4 of the LAR, the licensee described the ASME Section XI, Subsection IWE Program, and provided a summary of the results of containment inspections conducted for both units during refueling outages in 2013 through 2016. During RFO 1R26, inspections of the liner noted minor issues such as intermittent scratches and flaking paint to the exposed liner plate and new sections of moisture barrier were installed in several locations. The

licensee examined the moisture barriers used in the expansion joints located radially around the containment floor between the containment wall and the steam generator and reactor coolant pump platforms in response to NRC Information Notice (IN) 2014-07, "Degradation of Leak-Chase Channel Systems for Floor Welds of Metal Containment Shell and Concrete Containment Metallic Liner," May 5, 2014 (ADAMS Accession No. ML14070A114). Several areas of degraded moisture barrier were identified along with missing sealant at four locations, standing water, and varying levels of moisture were noted in most of the locations. All accessible floor covers were removed to view the pipe cap-to-pipe connection which serves as the moisture barrier. Several pipe caps were corroded which prompted inspection of additional locations. Ultrasonic testing (UT) measurements were performed on the inaccessible portion of the containment liner plate and found acceptable. In addition to the UT examinations, the licensee performed numerous visual inspections of the leak-chase channels and the issues found were evaluated to be acceptable. The licensee stated that beginning with the RFO 2R23 and 1R26 outages, the leak chase channels were included as part of the IWE scope, but were not included prior to the spring of 2012 since these items were not recognized as a potential location for moisture intrusion to the inaccessible portions of the liner plate. Subsequent to the issuance of the IN in May 2014, FNP added the leak-chase channels to the IWE Program as moisture barriers.

NRC Staff Evaluation

The NRC staff noted that no indications of significant degradation were identified in past IWE inspections. The NRC staff finds the licensee is properly implementing the ASME Section XI, Subsection IWE Program, which provides reasonable assurance that the structural integrity of the liner will be maintained if the ILRT frequency is extended as requested, and is, therefore, acceptable.

3.2.3.2 ASME Section XI, Subsections IWL

The licensee also described the ASME Section XI, Subsection IWL Program for both units and provided a summary of the inspections conducted in 2012 and 2015 which included containment coatings, general visual inspection of containment concrete surfaces, and containment structure post-tensioning system. The frequency of the concrete containment examinations for each unit is performed every 5 years in accordance with IWL-2410(a). The examinations required by IWL-2522 and IWL-2523 for a particular unit are performed every 10 years. The general visual concrete inspection indications included minor passive cracking, grease leaks, flaking paint and efflorescence, or rust bleeding, which were all dispositioned by the responsible engineer as acceptable; and no abnormal degradation of containment structures' post-tensioning system was identified.

NRC Staff Evaluation

The NRC staff noted that the indications found in past inspections were acceptable per the IWL acceptance criteria, and the indications identified no significant concrete spalling or loss of material in the associated reinforcing steel. The NRC staff finds the licensee is properly implementing the ASME Section XI, Subsection IWL Program, which provides reasonable assurance that the structural integrity of the concrete containment will be maintained if the ILRT frequency is extended as requested.

3.2.3.3 License Renewal Aging Management Programs

As discussed in the LAR, renewed operating licenses for FNP Units 1 and 2 were issued on May 12, 2005, extending the original licensed operating term by 20 years. FNP entered the period of extended operation for Unit 1 on June 26, 2017, and will enter the period for extended operation on April 1, 2021, for Unit 2. License renewal Aging Management Programs (AMPs) such as the containment ISI Program and 10 CFR 50, Appendix J, Option B, which are part of the supporting basis of the LAR, are also AMPs at FNP.

NRC Staff Evaluation

The continued implementation of applicable 10 CFR 50.55a requirements provides reasonable assurance that the aging effects will be managed such that systems and components within the scope of the program will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation. The Appendix J Program, along with the ASME Section XI, IWE and IWL programs, are used to detect initiation of aging degradation of containment at FNP.

The NRC staff finds that the licensee has adequately implemented the containment ISI program to periodically examine, monitor, and manage the condition of its containment structures. The results of past containment concrete and liner visual inspections demonstrate acceptable performance of the containment and demonstrate that the structural integrity of the containment structure is adequate. Thus, the NRC staff finds that there is reasonable assurance that the containment structural integrity will continue to be maintained if the current Type A interval is extended from 10 to 15 years.

3.2.4 NRC Staff Conclusion

Based on the above, the NRC staff concludes that the requested change is within the regulatory requirements and guidance as described in Section 2.0 of this SE. The licensee has implemented an adequate containment ILRT and Containment ISI program and supplemental inspections to periodically examine, monitor, and manage age-related degradation of FNP, Units 1 and 2, primary containments. The results of the past ILRTs and the effectiveness of the containment ISI programs demonstrate that the structural and leak-tight integrity of the primary containment is managed adequately, and will continue to be maintained without undue risk to public health and safety if the current ILRT interval is extended from 10 years to 15 years.

3.5 Probabilistic Risk Assessment (PRA) Review

Background

NEI 94-01, Revision 3-A, Section 9.2.3.1, "General Requirements for ILRT Interval Extensions beyond Ten Years," states that plant-specific confirmatory analyses are required when extending the Type A ILRT interval beyond ten years. Section 9.2.3.4, "Plant-Specific Confirmatory Analyses," states that the assessment should be performed using the approach and methodology described in EPRI TR 1009325, Revision 2-A[1], "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals." The analysis is to be performed by the

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1 EPRI TR-1009325, Revision 2-A, is also identified as EPRI TR-1018243. This report is publicly available and can be found at www.epri.com by typing "1018243" in the search field box.

licensee and retained in the plant documentation and records as part of the basis for extending the ILRT interval.

In NRC staff's final SE for NEI 94-01, Revision 2, the NRC staff found the methodology in EPRI TR-1009325, Revision 2, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals, Final Report," August 2007 (ADAMS Accession No. ML072970208) is acceptable for referencing by licensees proposing to amend their TSs permanently to extend the ILRT interval to 15 years, provided the following four conditions are satisfied. These four conditions, titled, "Limitations and Conditions for EPRI Report No. 1009325, Revision 2" are located in Section 4.2 of NEI 94-01, Revision 2, and summarized below.

Condition 1 – PRA Technical Adequacy

> The licensee submits documentation indicating that the technical adequacy of their PRA is consistent with the requirements of RG 1.200 [Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-informed Activities," dated March 2009 (ADAMS Accession No. ML090410014)] relevant to the ILRT extension application.
>
> NRC staff notes that additional application specific guidance on the technical adequacy of a PRA used to extend ILRT intervals is provided in the SER for EPRI TR-1009325, Revision 2.

Condition 2 - Estimated Risk Increase, states, in part:

> The licensee submit documentation indicating that the estimated risk increase associated with permanently extending the ILRT surveillance interval to 15 years is small and consistent with the clarification provided in Section 3.2.4.6[2] of this SE [EPRI TR-1009325, Revision 2.]

Condition 3 - Leak Rate for the Large Pre-Existing Containment Leak Rate Case, states:

> The methodology in EPRI TR-1009325, Revision 2, is acceptable except for the calculation of the increase in expected population dose (per year of reactor operation). In order to make the methodology acceptable, the average leak rate for the pre-existing containment large leak accident case (accident case 3b) used by licensees shall be 100 L_a instead of $35L_a$.

Condition 4 to reference EPRI Report No. 1009325, Revision 2, states:

> A LAR is required in instances where containment over-pressure is relied upon for ECCS [emergency core cooling system] performance.

3.5.2 Plant-Specific Risk Assessment

As required per NEI 94-01, Revision 3-A, the licensee performed a plant-specific risk assessment of the effect on risk metrics for extending the Type A containment ILRT interval from 10 years to 15 years at FNP, Units 1 and 2. This assessment was provided by the

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2 The SE for EPRI TR-1009325, Rev. 2, indicates that the clarification regarding small increases in risk *is provided* in Section 3.2.4.5, however, the clarification is actually provided in Section 3.2.4.6.

licensee in Attachment 1 of the licensee's submittal dated November 15, 2016. Additional information associated with the Farley risk assessment was provided by the licensee in SNC's supplemental letters dated June 22, 2017, September 11, 2017, October 12, 2017, and February 9, 2018.

In Section 1.1 of Attachment 1 to the LAR, the licensee stated that the plant-specific risk assessment for Farley follows the guidance in:

- Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J, NEI 94-01, Revision 3-A, July 2012.

- Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals, EPRI, Palo Alto, CA, EPRI TR-104285, August 1994.

- Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals, Revision 4, developed for NEI by EPRI and Data Systems and Solutions, November 2001.

- RG 1.174, Revision 2, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," May 2011 (ADAMS Accession No. ML 100910006).

- Response to Request for Additional Information Concerning the License Amendment Request for a One-Time Integrated Leakage Rate Test Extension, Letter from Mr. C. H. Cruse (Calvert Cliffs Nuclear Power Plant (CCNP)) to NRC Document Control Desk, Docket No. 50-317, March 27, 2002 (ADAMS Accession No. ML020920100).

- Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals, EPRI, Palo Alto, CA, EPRI TR-1009325 Revision 2-A, 2008.

3.5.3 Four Conditions to reference EPRI Report No. 1009325, Revision 2

The NRC staff reviewed and performed independent confirmatory calculations to ensure the licensee has met the four "Limitations and Conditions for EPRI Report No. 1009325, Revision 2." The NRC staff evaluation is provided below.

3.5.3.1 Condition 1 - PRA Technical Adequacy

The first condition stipulates that the licensee submits documentation indicating that the technical adequacy of their PRA is consistent with the requirements of RG 1.200 relevant to the ILRT extension application.

Internal Events

The PRA quality (including the technical adequacy of the PRA), as it relates to the ILRT extension application, is described in Section 3.2.4.1 of the SE to EPRI TR-1009325, Revision 2 (ADAMS Accession No. ML081140105), and states, in part:

> Licensee requests for a permanent extension of the ILRT surveillance interval to 15 years pursuant to NEI TR 94-01, Revision 2, and EPRI Report No. 1009325, Revision 2, will be treated by NRC staff as risk-informed license amendment requests. Consistent with information provided to industry in Regulatory Issue Summary 2007-06, "Regulatory Guide 1.200 Implementation"... the NRC staff will expect the licensee's supporting Level 1/LERF [large early release frequency] PRA to address the technical adequacy requirements of RG 1.200, Revision 1... Any identified deficiencies in addressing this standard shall be assessed further in order to determine any impacts on any proposed decreases to surveillance frequencies. If further revisions to RG 1.200 are issued which endorse additional standards, the NRC staff will evaluate any application referencing NEI TR 94-01, Revision 2, and EPRI Report No. 1009325, Revision 2, to examine if it meets the PRA quality guidance per the RG 1.200 implementation schedule identified by the NRC staff.

The NRC staff also stated that Capability Category I of the ASME PRA standard shall be applied as the standard for assessing PRA quality for ILRT extension applications, as approximate values of core damage frequency (CDF) and LERF and their distribution among release categories are sufficient to support the evaluation of changes to ILRT frequencies.

In Attachment 1, Appendix A of the LAR, the licensee stated that the Farley PRA model used is the Farley Units 1 and 2 Internal Events PRA Revision 9, Version 3.

The licensee stated that their Internal Events PRA models have undergone several peer reviews including:

1) An independent PRA peer review conducted under the auspices of the Westinghouse Owners Group (WOG) in 2001, following the industry PRA peer review process. This peer review included an assessment of the PRA model maintenance and update process.

2) In 2005, a gap analysis was performed against the available version of the ASME PRA Standard and Regulatory Guide 1.200, Revision 0.

3) Another Farley Unit 1 PRA Peer Review was performed in March of 2010 using the NEI 05-04 process, the 2009 version of the ASME/ANS PRA standard, and Regulatory Guide 1.200, Revision 2. The purpose of this review was to provide a method for establishing the technical adequacy of the PRA for the spectrum of risk-informed plant licensing applications for which the PRA may be used. This Peer Review was a full-scope review of Revision 9, Version 1, of the Farley Unit 1 PRA model. From this review there were 17 SRs (Supporting Requirements) that were found to be "Not Met". These 17 SR items have been addressed and closed and were attached for review in Appendix A to the LAR.

In SNC letter dated June 22, 2017, the licensee stated no self-assessment or peer review has been performed for internal events (including internal flooding) and FPRA models since their NFPA 805 submittal. Therefore, no new findings have been generated for these models since the NFPA 805 submittal. Consequently, the governing Peer Review for internal events (and internal flooding) is that from 2009.

The licensee also stated in their letter dated June 22, 2017, that some peer review findings for internal events (including internal flooding) PRA were characterized as "pending" in the NFPA 805 submittal but have subsequently been dispositioned. Therefore, all findings and their resolution were provided in their response to RAI-4 in the June 22, 2017, supplement.

The NRC staff reviewed the peer review findings and dispositions provided by the licensee to ensure the peer review findings addressed the licensee's treatment of dependency between multiple human actions within cutsets. The NRC staff found most of the peer review dispositions to be acceptable except for the dispositions of peer review findings HR-G7-01 and HR-G7-02, for which staff requested additional information. The NRC staff asked the licensee if a human error probability floor value was used for joint human error probabilities (JHEPs) within a cutset in both the Farley internal events PRA and the external events PRA because, without the use of a floor value, inappropriately low JHEPs within a cutset are possible.

In its letter dated, June 22, 2017, the licensee provided additional information regarding these peer review findings/dispositions. For HR-G7-01 and 02, the licensee stated that a JHEP floor value of 1.0E-6 was used for HRA Dependency Analysis in the Farley internal events PRA, and that a JHEP floor value of 1.0E-5 was used for their Fire PRA. The NRC staff finds that these are appropriate JHEP floor values to be used within cutsets in both their internal and fire PRAs, and that this approach is conservative and appropriate for this application. Therefore, the NRC staff finds that the disposition of HR-G7-01 and HR-G7-02 is acceptable.

The NRC staff also questioned F&O IE-A10-02 regarding a failure of the Farley Service Water (SW) pond dam. The NRC staff questioned if the original dam failure analysis had been updated and if the criteria that SNC used for screening out this event from their internal events PRA still applied, including the quantitative basis for this screening given it was questioned by the Peer Review as a Finding.

In its letter dated June 22, 2017, the licensee provided details of how the F&O for this service water pond dam failure had been closed out. The licensee stated that the SW pond failure analysis had not been updated using any new approaches. The licensee provided excerpts from the Farley Internal Events PRA Peer Review F&O resolution document, F-RIE-IEIF-U00-12, in response to the NRC staff's question.

The licensee also provided information on the SW pond licensing basis along with a re-calculation of the failure frequency of a random dam failure using generic information as well as information supporting the reduction in failure frequency. However, the NRC staff reviewed this information, and agreed with the original F&O's concern that the licensee is reducing the dam failure frequency by a factor of 100 based on qualitative considerations already credited to assign the lowest generic dam failure frequency to the scenario. As a result, the NRC requested that the licensee perform a sensitivity analysis incorporating the dam failure event at the generic frequency (without the factor of 100 reduction) to demonstrate that incorporating this event into the internal events would have negligible effect upon the acceptability of the risk metrics for this application. The NRC staff requested this additional information via RAI 6, in NRC email dated August 3, 2017 (ADAMS Accession No. ML 17219A001).

In its letter dated September 11, 2017, the licensee responded by including a sensitivity study that showed how including the SW pond dam failure would affect their internal events PRA and subsequent metrics for this application. The sensitivity showed that the increase in risk by including the SW pond dam failure resulted in a CDF of 3.49E-7/yr for each unit and LERF

of 4.53E-9/yr (for one test in 15 years) per unit. These results were also included in the final RAI response letter dated February 9, 2018.

Additionally, although SNC initially did not credit the Westinghouse Generation III Reactor Coolant Pump (RCP) seal in the LAR. In the response to RAI 7, dated October 12, 2017, SNC stated that the revised values for Internal Events included credit for the Generation III RCP shutdown seals. The NRC staff asked SNC to confirm if the new values followed the August 23, 2017, Topical Report [TR] PWROG-140001-P, Revision 1, 'PRA Model for the Generation III Westinghouse Shutdown Seal" and the associated NRC Safety Evaluation including the Limitations and Conditions (ADAMS Package Accession No. ML17200A116).

In its letter dated February 9, 2018, SNC's CDF and LERF values comply with the Topical Report PWROG-140001-P, Revision 1 with the exception of Conditions 2 and 4. SNC stated that its Internal Events PRA has been revised since the October 12, 2017, submittal to include Condition 2, but does not yet include Condition 4. Condition 4 required licensees with RCP model 93A installed to incorporate the shutdown seal bypass mode into their PRA. However, based on a sensitivity performed using the latest Internal Events model, Condition 4 results in an increase in CDF of only 0.07 percent and increase in LERF of only 0.01 percent. Additional information on the impact of including credit for Westinghouse Generation III Shutdown Seal can be found in Section 3.5.3.2 of this SE.

External Events

In addition to the Internal Events PRA models used for their ILRT Risk Assessment, Farley has Fire PRA (FPRA) models for Units 1 and 2 which were developed for Risk Informed applications including NFPA 805. These models underwent a RG 1.200, Revision 2, peer review against the ASME PRA Standard, conducted by the Pressurized Water Reactor Owner's Group (PWROG) in October 2011 in accordance with NEI 07-12. The peer review concluded that the methodologies used in development of the Farley FPRA models were appropriate and sufficient to satisfy ASME/ANS Standard RA-Sa-2009 as endorsed by Revision 2 of RG 1.200.

In SNC letter dated June 22, 2017, the licensee stated no self-assessment or peer review has been performed for FPRA models since their NFPA 805 submittal. Therefore, no new findings have been generated for these models since the NFPA 805 submittal. The disposition of the FPRA findings has not changed as a result of the processing of the NFPA-805 submittal, and therefore, the reference made to the NFPA 805 submittal remains valid for the FPRA model.

In Section 3.2.4.2 of the SE for NEI 94-01, Revision 2, and EPRI TR-1009325, Revision 2, the NRC staff states that:

> Although the emphasis of the quantitative evaluation is on the risk impact from internal events, the guidance in EPRI Report No. 1009325, Revision 2, Section 4.2.7, "External Events," states that: "Where possible, the analysis should include a quantitative assessment of the contribution of external events (e.g., fire and seismic) in the risk impact assessment for extended ILRT intervals."

In its letter dated November 15, 2016, the licensee stated that they have FPRA models for Units 1 and 2, which were developed for Risk Informed applications including National Fire Protection Association Standard 805 (NFPA 805). The FNP FPRA models underwent a RG 1.200, Revision 2, peer review against the ASME PRA Standard, conducted by the Pressurized Water

Reactor Owner's Group (PWROG) in October 2011 in accordance with NEI 07-12. The peer review concluded that the methodologies used in development of the FNP FPRA models were appropriate and sufficient to satisfy ASME/ANS PRA Standard RA-Sa-2009 (as clarified/qualified by Revision 2 of RG 1.200). The results of the FPRA peer review were included in the LAR submitted to the NRC for approval to transition to NFPA 805. FNP also stated that the values for the FPRA come from Table 3-1 of the Farley FPRA Summary Report Calculation PRA-BC-F-11-017, Revision 0. The Farley FPRA models credit pending modifications for NFPA 805 that will be fully implemented by the conclusion of the FNP, Unit 1, spring 2018 Refueling Outage (1R28).

There have been changes to FPRA methods since the safety evaluation (SE) was issued to FNP for NFPA 805. In particular, VEWFDS was given credit in the CDF and LERF calculation consistent with NFPA Standard 805 Frequently Asked Question (FAQ) 08-0046, "Incipient Fire Detection Systems." However, in NRC letter dated July 1, 2016 (ADAMS Accession No. ML16167A444), the NRC superseded the FAQ 08-0046 interim guidance with NUREG-2180, "Determining the Effectiveness, Limitations, and Operator Response for Very Early Warning Fire Detection Systems in Nuclear Facilities (DELORES-VEWFIRE)" (ADAMS Accession No. ML16343A058).

Therefore, in NRC letter dated March 15, 2017 (ADAMS Accession No. ML17058A113), as supplemented via NRC email dated August 3, 2017 (ADAMS Accession No. ML17219A001), the NRC staff asked, in RAI 5, if the fire CDFs and LERFs reflect changes to FPRA methods since the FNP NFPA-805 safety evaluation was issued. In this RAI, the NRC staff specifically requested SNC to address the credit taken in FAQ 08-0046; the changes in fire ignition frequencies and non-suppression probabilities in NUREG-2169, "Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database" (ADAMS Accession No. ML15016A069); and, possible increases in spurious operation probabilities in NUREG/CR-7150, "Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE)," Volume 2, "Expert Elicitation Exercise for Nuclear Power Plant Fire-Induced Electrical Circuit Failure" (ADAMS Accession No. ML14141A129).

The licensee responded to RAI 5 in letters dated October 12, 2017, and in February 9, 2018. In its responses, the licensee provided a description of sensitivity studies that were performed using the new guidance. The licensee provided revised PRA values for LERF and CDF that are discussed further in Section 3.5.3.2 of this SE.

Regarding seismic events, in its submittal dated November 15, 2016, Table 6-2, "Farley Internal and External Events Summary," the licensee provided CDF and LERF estimates. The NRC staff reviewed this application and requested that the licensee provide additional information (RAI 7) concerning the Table 6.2 calculations of seismic values for LERF and CDF as cited in the NFPA 805 Safety Evaluation, as these were higher than the values cited in this LAR, and confirm that any increase in the risk metrics as a result of the recalculation using the values previously cited for NFPA 805 does not change the justification for meeting the acceptance criteria.

In its response to RAI 7 on October 12, 2017, the licensee provided a revised Table 6-2 with new values that were based on a new update of those approved in the NFPA 805 Transition license amendment dated March 10, 2015 (ADAMS Accession No. ML14308A048), incorporating the Seismic Hazard Reevaluation and Screening for Risk Evaluation

report NL-14-0342. A final revised Table 6-2 was submitted in the February 9, 2018, RAI 7 supplemental response.

Regarding other external events, the licensee showed in Table 6-2 of its submittal dated November 15, 2016, that other external events were not a significant contributor to risk, such that these events could be screened out.

SNC has made further revisions to both the Internal Events and Fire PRA models since the October 12, 2017, RAI response that corrected an error in the exponent of the Seismic LERF value reported in the RAI response from October 12th. This is reflected in Section 3.5.3.2 of this SE.

Summary of Internal and External Events

In summary, the licensee has evaluated its internal events PRA against the currently endorsed ASME PRA standard (i.e., ASME/ANS RA-Sa-2009) and RG 1.200, Revision 2; evaluated the findings developed during the peer review of its internal events PRA for applicability to the ILRT interval extension; addressed the findings or evaluated their impact; and included a quantitative assessment of the contribution of external events. The staff reviewed the internal events and FPRA peer review findings and, based on the information provided in the application and subsequent RAI responses, the dispositioned findings have been adequately addressed for this application. Furthermore, the NRC staff finds that the impact from external events was considered appropriately.

NRC Staff Conclusion

Based on the above, the NRC staff concludes that the PRA used by the licensee is of sufficient technical adequacy to support the evaluation of changes to ILRT frequency. Therefore, NRC staff concludes that Condition 1 in the "Limitations and Conditions for EPRI Report No. 1009325, Revision 2" has been met.

3.5.3.2 Condition 2 - Estimated Risk Increase

The second condition stipulates that the licensee submit documentation indicating that the estimated risk increase associated with permanently extending the ILRT interval to 15 years is small, and consistent with the guidance in RG 1.174, Revision 2, and the clarification provided in Section 3.2.4.5 of the NRC SER for NEI 94-01, Revision 2-A, and EPRI TR-1009325, Revision 2. Specifically, a small increase in population dose should be defined as an increase of no more than 1.0 person-rem per year or one percent of the total population dose, whichever is less restrictive. In addition, a small increase in conditional containment failure probability (CCFP) should be defined as a value marginally greater than that accepted in previous one-time 15-year ILRT extension requests. This would require that the increase in CCFP be less than or equal to 1.5 percent. Additionally, for plants that rely on containment over-pressure for net positive suction head (NPSH) for ECCS injection, both CDF and LERF will be considered in the ILRT evaluation and compared with the risk acceptance guidelines in RG 1.174. As discussed in the response to RAI 9 dated June 22, 2017, FNP, Units 1 and 2, do not rely on containment over-pressure for ECCS NPSH. Thus, the associated risk metrics for this application include LERF, population dose, and CCFP.

In its submittal dated November 15, 2016, Section 4.2.4, "Population Dose Estimate Methodology," the licensee compared FNP with a reference plant, Surry. The NRC staff

questioned if the weather conditions around the two sites (Surry and Farley) are sufficiently comparable to render the scaling approach conservative based on power level, leakage and population. In SNC letter dated June 22, 2017, the licensee responded that the weather conditions around the two sites are sufficiently comparable because both Farley and Surry are in areas considered by climatologists to be Humid Subtropical Climates. Furthermore, since the EPRI methodology focuses on population density, power level, and allowed leakage as the relevant driving factors, weather can be considered secondary to the population density, power level and allowed plant leakage. The NRC staff finds the response was acceptable for this ILRT application.

The licensee reported the results of the plant-specific risk assessment. The reported risk impacts are based on a change in test frequency from three tests in 10 years (the test frequency under 10 CFR 50 Appendix J, Option A) to one test in 15 years.

These are summarized in Section 7.0 of Attachment 1 to the November 15, 2016, LAR, and are consistent with the RAI responses from June 22, 2017, September 11, 2017, October 12, 2017, and February 9, 2018.

Changes During the RAI Process

The estimated increases in risk metrics were originally presented in Attachment 1 of the November 15, 2016, application. However, the NRC staff requested additional information regarding these values. SNC provided revised estimates in RAI 7 in the October 12, 2017, and February 9, 2018, supplements.

The revised numbers were obtained by SNC by performing various sensitivities and recalculations requested during the RAI process. SNC stated in their RAI response of October 12, 2017:

> SNC has performed model updates and has recalculated Table 6-2 which lists the Farley CDF and LERF values for each internal and external event type that are used to determine the potential impact from the External Events contribution. The values for the Internal Events PRA come from the latest model of record (Revision 10, Version 1) for each unit which includes credit for the Generation III RCP shutdown seals. The values for the Fire Events PRA come from the response to RAI 5 above. The Seismic values come from an update of those approved in the NFPA 805 Transition license amendment (ADAMS Accession No. ML14308A048), incorporating the Seismic Hazard Reevaluation and Screening for Risk Evaluation report provided to the NRC previously in NL-14-0342 in March 2014. The values for a new initiating event, Loss of SW Pond Dam, come from the calculation of impact on CDF and LERF due to loss of the Service Water pond dam with credit given for the River Water System – these values were submitted to the NRC on September 11, 2017 in response to RAI 6. The recalculated table 6-2 follows."

As mentioned in Section 3.5.3.1 of this SE, the NRC staff requested that SNC supplement its RAI 7 response in the letter dated October 12, 2017, because the CDF and LERF values provided in the October 12, 2017, supplement included credit for the Westinghouse Generation III RCP seals. The NRC staff asked SNC to confirm if the new values followed the August 23, 2017, Topical Report [TR] PWROG-140001-P, Revision 1, 'PRA Model for the Generation III

Westinghouse Shutdown Seal," and the associated NRC Safety Evaluation including the Limitations and Conditions.

In its letter dated February 9, 2018, SNC also submitted revised tables and information showing the latest metrics for acceptance.

Seismic CDF Sensitivity

The NRC staff did not accept the submitted values for seismic CDF and instead performed a sensitivity for Seismic CDF. This is because, based on the values submitted on the October 12 response, the NRC staff observed that, relative to the values approved in the NFPA-805 License Amendment (ADAMS Accession No. ML14308A048), the seismic CDFs at each unit are reduced by approximately a factor of four, inconsistent with the approximate reduction by only a factor of two for the internal events CDF at each unit. Therefore, the seismic CDFs were rescaled by the NRC staff via bounding calculations to determine if the effect on the conclusions would be minimal. In this case, the seismic CDFs were scaled in the same proportions as the changes in the internal events CDFs (since the seismic PRA should be based on the internal events PRA), i.e., reductions by 56% at Unit 1 and 53% at Unit 2. The resulting seismic CDFs become 7.61E-6/yr for Unit 1 and 8.13E-6/yr for Unit 2. As shown below in Table 1 (with the licensee-submitted values in parentheses), using these sensitivity bounds does not adversely impact the conclusions for the application.

Using these sensitivity values for seismic CDF resulted in total CDFs of 9.99E-5/yr for Unit 1 and 9.61E-5/yr for Unit 2. This is shown in the table below. The values in this table represent the final results based on the numbers submitted by SNC in the February 9, 2018, supplement response, with the NRC staff substituted values for Seismic CDF (SNC's submitted numbers in parenthesis).

Farley Internal and External Events Summary				
Event Type	Farley Unit 1		Farley Unit 2	
	CDF (per/year)	LERF (per/year)	CDF (per/year)	LERF (per/year)
Internal Events	8.90E-06	9.76E-08	8.76E-06	7.93E-08
Loss of SW Dam	3.49E-07	4.53E-09	3.49E-07	4.53E-09
Fire Events	8.35E-05	4.21E-06	7.89E-05	4.51E-06
Seismic	7.61E-6 (4.51E-06)	2.07E-06	8.13E-6 (4.51E-06)	2.07E-6
Other External Risk	Screened Out			
Total	9.99E-05 (9.73E-05)	6.38E-06	9.61E-05 (9.25E-05)	6.66E-06

Using these new values for CDF obtained from the table above, the NRC staff extrapolated a revised table to show the estimated total LERF shown below (SNC's submitted numbers in parenthesis).

Farley Estimated Total LERF Including External Events Impact				
Case	3b frequency (3 per 10 year test)	3b frequency (1 per 10 year test)	3b frequency (3 per 15 year test)	LERF Increase (3 per 10 year test to 1 per 15 year)
Unit 1 Internal Events Contribution	2.03E-08	6.74E-08	1.01E-07	8.10E-08
Unit 1 Total Contribution including External Events	2.15E-07 (2.09E-7)	7.17E-7 (6.96E-07)	1.07E-6 (1.05E-06)	8.60E-07 (8.36E-07)
Unit 2 Internal Events Contribution	2.00E-08	6.65E-08	9.98E-08	7.99E-08
Unit 2 Total Contribution including External Events	2.06E-07 (1.97E-07)	6.86E-07 (6.58E-07)	1.03E-06 (9.87E-07)	8.23E-07 (7.90E-07)

When SNC responded on February 9, 2018, with a supplement to RAI 7, the licensee provided revised values for increases in population dose and conditional containment failure probability:

> "[T]he ILRT interval extension risk assessment values reported in the LAR (e.g., population dose increase and increase in conditional containment failure probability) were re-calculated using the latest version of the internal events model which includes Condition 2, and modified to incorporate Condition 4 to show that the final results remain in the acceptable range."

Summary of Acceptance Metrics

The final estimates of the metrics involving changes in risk associated with the application are shown in the below table. Included are the Licensee's submitted results from their letter dated February 9, 2018, in parenthesis prior to the staff's preceding "sensitivity" bounding analysis.

Risk Metrics with NRC Staff "Sensitivity" Bounds	Unit 1	Unit 2
Increase in total LERF due to change in test frequency from 3-in-10 to 1-in-15 years, including external events	8.60E-07/yr (8.36E-7/yr)	8.23E-7/yr (7.90E-7/yr)
Total LERF including external events	6.38E-6	6.66E-6
Increase in Conditional Containment Failure Probability due to change in test frequency from 3-in-10 to 1-in-15 years	0.91%	0.92%
Increase in population dose due to change in test frequency from 3-in-10 to 1-in-15 years	5.02E-03 person-rem/yr	4.94E-03 person-rem/yr

These results include the seismic sensitivity bounding analysis performed by the staff and show that the values for the increases in the relevant risk metrics for this application would not exceed acceptance thresholds. Regarding the increases in conditional containment failure probability and population dose in the above table, SNC did not revise the original tables contained in Attachment 1, Section 5, of their submittal dated November 15, 2016, that would show how these values were calculated for the final results. Nonetheless, the NRC staff concludes that, since Internal Events total CDF for each unit has dropped by roughly half of what was originally submitted, the results remain bounding and a full re-submittal of section 5 is not necessary.

As summarized in Section 7 of Attachment 1 to the November 15, 2016 submittal, and the subsequent RAI responses from February 9, 2018, the following are conclusions regarding the assessment of the plant risk are associated with permanently extending the Type A ILRT test frequency to once in fifteen years:

- Regulatory Guide 1.174 provides acceptance criteria for increases in CDF and LERF resulting from a risk-informed application. The relevant criterion for this application is LERF. When the calculated increase in LERF is "very small", which is taken as being less than 1.0E-7 per reactor-year, the change will be generally considered acceptable irrespective of the plant's LERF value. When the calculated increase in LERF is in the range of 1.0E-7 per reactor-year to 1.0E-6 per reactor year, ["small"], the applications will be considered acceptable only if the total plant LERF is less than 1.0E-5 per reactor-year.

- The licensee's assessment includes the impact from External Events. In this case, the total class 3b contribution to LERF including External Events was estimated as 1.07E-06/yr for Unit 1 and 1.03E-6/yr for Unit 2 of FNP, including the staff sensitivity evaluation. In addition, based on Table 3 and including the staff sensitivity evaluation, the total LERF based on the inclusion of external events impacts is 6.38E-6/yr for Unit 1 and 6.66E-6/yr for Unit 2. This is below the RG 1.174 acceptance criterion for total LERF of 1E-05/yr for a "small" change, therefore satisfying both the incremental and absolute expectations with regard to the RG 1.174 LERF metric.

- Based on the "Risk Metrics with NRC Staff "Sensitivity" Bounds" table, the increase in LERF resulting from a change in the Type A ILRT test interval from three in ten years to one in fifteen years, is estimated as 8.60E-7/yr for Unit 1 and 8.23E-7/yr for Unit 2, using the EPRI guidance as written. These estimated changes in LERF for Farley 1 and Farley Unit 2 are within the Reg. Guide 1.174 "small" risk increase criterion, for which total LERF must be below 1.00E-5/yr for acceptability.

- The change in Type A test frequency to once per fifteen years, measured as an increase to the total integrated plant risk [population dose] for those accident sequences influenced by Type A testing based on internal events PRA, is 5.02E-03 person-rem/yr for Unit 1, and 4.94E-03 person-rem/yr for Unit 2. EPRI Report No. 1009325, Revision 2-A states that a very small population dose is defined as an increase of $\leq$1.0 person-rem per year or $\leq$1 % of the total population dose, whichever is less restrictive for the risk impact assessment of the extended ILRT intervals. This is consistent with the NRC Final Safety Evaluation for NEI 94-01 and EPRI Report No. 1009325.

- The increase in the conditional containment failure probability from the three-in-ten-year interval to a permanent one time in fifteen year interval is 0.91% for FNP Unit 1 and 0.92% for Unit 2. EPRI Report No. 1009325, Revision 2-A states that increases in CCFP of $\leq$1.5 percent are very small. This is consistent with the NRC Final Safety Evaluation for NEI 94-01 and EPRI Report No. 1009325. Therefore, NRC staff finds this increase to be very small.

NRC Staff Conclusion

Based on the risk assessment results, the NRC staff concludes that the increases in LERF are small and consistent with the acceptance guidelines of RG 1.174, which include the total LERFs. The increases in the total population dose and the magnitudes of the change in the CCFP for the proposed change are very small and supportive of the LAR. The defense-in-depth philosophy is maintained as the independence of barriers will not be degraded as a result of the requested change, and the use of the three quantitative risk metrics collectively ensures that the balance between prevention of core damage, prevention of containment failure, and consequence mitigation is preserved. Therefore, NRC staff concludes that Condition 2 in the "Limitations and Conditions for EPRI Report No. 1009325, Revision 2" has been met.

3.5.3.3 Condition 3 - Leak Rate for the Large Pre-Existing Containment Leak Rate Case

The third condition stipulates that in order to make the methodology in EPRI TR-1009325, Revision 2, acceptable, the average leak rate for the pre-existing containment large leak rate accident case (i.e., accident case 3b) used by the licensees shall be $100L_a$ instead of 35 L_a.

As noted by the licensee in Section 3.3.1 of the LAR, the methodology in EPRI TR-1009325, Revision 2, incorporates the use of $100L_a$ as the average leak rate accident case, and this value has been used in the Farley plant-specific risk assessment. Therefore, NRC staff concludes that Condition 3 in the "Limitations and Conditions for EPRI Report No. 1009325, Revision 2" has been met.

3.5.3.4 Condition 4 - Applicability if Containment Over-Pressure is Credited for ECCS Performance

The fourth condition stipulates that in instances where containment over-pressure is relied upon for ECCS performance, a LAR is required to be submitted. In its letter dated June 22, 2017, the licensee stated that Plant Farley Units 1 and 2 do not rely on containment overpressure for ECCS performance. Therefore, NRC staff concludes that Condition 4 in the "Limitations and Conditions for EPRI Report No. 1009325, Revision 2" is not applicable to this application.

3.5.4 PRA NRC Staff Conclusion

Based on the above, the NRC staff concludes that the licensee has met the four "Limitations and Conditions for EPRI Report No. 1009325, Revision 2." The NRC staff concludes that the PRA used by the licensee is sufficient and that the risk impact for extending the integrated leak rate testing intervals is consistent with the acceptance guidelines of RG 1.174.

3.6 NRC Staff Conclusion

Based on the preceding regulatory and technical evaluations, the NRC staff concludes that the licensee has adequately implemented its primary containment leakage rate testing program

consisting of ILRT and LLRT. The results of the recent ILRTs and the LLRT (Type B and Type C tests) combined totals demonstrate acceptable performance and support a conclusion that the structural and leak-tight integrity of the primary containment vessel is adequately managed and will continue to be periodically monitored and managed effectively. The NRC staff concludes that the licensee has addressed the NRC conditions to demonstrate acceptability of adopting TR NEI 94-01, Revision 3-A, and the limitations and conditions identified in the staff SE incorporated in TR NEI 94-01, Revision 2-A. The NRC staff concludes that the risk impact for extending the integrated leak rate testing intervals is consistent with the acceptance guidelines of RG 1.174. Therefore, the NRC staff concludes that the proposed changes to FNP TS 5.5.17 regarding the primary containment leakage rate testing program are acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of Alabama official was notified of the proposed issuance of the amendments on November 20, 2017. On November 20, 2017, the State confirmed that the State of Alabama had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on January 3, 2017 (82 FR 161). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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Date: February 28, 2018

SUBJECT: JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2 – ISSUANCE OF AMENDMENTS TO REVISE TECHNICAL SPECIFICATION 5.5.17, "CONTAINMENT LEAKAGE RATE TESTING PROGRAM" (CAC NOS. MF8844, MF8845; EPID NO. L-2016-LLA-0015) DATED FEBRUARY 28, 2018

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